MNSR transient analysis using the RELAP5/Mod3.2 code

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Abstract

To support the safe operation of the Miniature Neutron Source Reactor (MNSR), a thermo-hydraulic transient model using the RELAP5/Mod3.2 code was simulated. The model was verified by comparing the results with the measured and the previously calculated data. The comparisons consisted of comparing the MNSR parameters under normal constant power operation and reactivity insertion transients. Reactivity Insertion Accident (RIA) for three different initial reactivity values of 3.6, 6.0, and 6.53 mk have been simulated. The calculated peaks of the reactor power, fuel, clad and coolant temperatures in hot channel were calculated in this model. The reactor power peaks were: 103 kW at 240 s, 174 kW at 160 s and 195 kW at 140 s, respectively. The fuel temperature reached its maximum value of 116 °C at 240 s, 124 °C at 160 s and 126 °C at 140 s respectively. These calculation results ensured the high inherently safety features of the MNSR under all phases of the RIAs.

Keywords:
MNSR
RELAP
MCNP
Transient analyses
Safety

1. Introduction

The MNSRs have been designed and manufactured by the China Institute of Atomic Energy since the mid-1980s. A total of nine MNSRs have been built: four in China and one each in Pakistan (1989), Islamic Republic of Iran (1994), Ghana (1995), Syrian Arab Republic (1996), and Nigeria (2004). MNSRs are used mainly for neutron activation analysis, training and education [1]. Several transient experiments and calculations were performed through the insertion of large reactivities to demonstrate the self-limiting power excursion features of MNSRs [2–6].

Recently, many computational tools have been used in several studies. For example: RELAP5 (Reactor Excursion and Leak Analysis Program), ATHLET (Analysis of Thermal-hydraulics of leaks and Transients), PARET (Program for analysis of reactor transient), COBRA (Coolant Boiling in Rod Arrays), PARCS (Purdue advanced reactor core simulator), MERSAT (Model for Evaluation of Reactor Safety and Analysis of Thermal-Hydraulics), and the coupled PARCS/RELAP5.

However, the safety analysis of the reactor requires validated TH (thermal-hydraulic) code. It implies the integrated neutronic- thermo-hydraulic analysis to cover the various phases of reactor operations during the reactor life cycles. Robust tool should be provided by the comprehensive analysis to demonstrate the safety of research reactor according to its design limits. The reactivity insertion accidents of 3.6, 6.0, and 6.53 mk were analyzed using RELAP5/Mod3.2 code to prove the high inherently safety features of the MNSR under all phases of these reactivity insertion accidents and to verify the self-limiting power excursion behavior of the MNSR under the reactivity insertion accidents.

2. Methodology

2.1. Reactor description

The Syrian Research Reactor is the commercial version of the MNSR and belongs to the class of tank-in-pool type reactors [7]. Thermal power is rated at 30 kW with the corresponding thermal neutron flux of 1.0 × 10^{12} n/cm².s. For the fresh core, the reactor cold excess reactivity was about 4 mk. Cooling of the reactor is conducted by natural convection using light water. Presently, the MNSR core consists of HEU (U–Al₄ alloyed) fuel rods arranged in ten concentric rings about the central control rod guide tube which houses the reactor's only control rod. The fuel enrichment is about 90% U²³⁵. Schematic diagrams of the MNSR are shown in Fig. 1. The reactor core contains 347 active fuel pins plus 3 dummy pins. The core is surround by the annulus Be reflector. Another Be reflector is noticed under the core. Adding a regulated thickness of beryllium shims to the top reactor tray can compensate the top axial neutron leakage. The core heat is removal by conduction method through the tank wall to the pool water.
2.2. MCNP calculations

The neutronic parameters of the MNSR such as: the delayed neutron fraction, reactivity coefficients and other parameters were calculated by the MCNP4C [8]. The MNSR real dimensions were constructed in the MCNP model as mentioned in the reactor safety analysis report [7]. Then, these parameters were used in the RELAP input file for conducting thermo-hydraulic calculations of HEU core at normal and transient conditions. The maximum allowed reactivity insertion in the MNSR is 4 mk. An additional of 2.5 mk may result from the filling of the inner irradiation site accident. A reactivity insertion accident of 3.6, 6.0, and 6.53 mk were analyzes using the RELAP code for the original HEU MNSR core. The reactivity transient of 5.5 mk was also conducted as well to compare with previously published results.

Calculations using the MCNP4C code were performed to obtain power shapes and reactivity feedback coefficient for the HEU core. Fig. 2 shows the geometry used for these calculations. Table 1 shows the kinetics parameters and temperature feedback coefficient obtained from these calculations.

2.3. RELAP5 model

Analysis of the MNSR with HEU fuel was performed using the RELAP5 code to determine the safety margins in the normal operation and RIA accidents. Our model using the RELAP5 code found out the hot channel for the highest power pin and its coolant, and the average channel for the remaining 347 pins and their coolant. The MNSR was divided in this model to many sections which consisted of: the reactor core (primary system) and the reactor pool (secondary system) which are thermally interconnected through the wall of reactor vessel [9].

2.3.1. Fuel model for heat transfer calculation

Fig. 3 shows schematic diagram of the coolant flow pattern and coolant channel for MNSR reactor. Cold water is drawn through the

<table>
<thead>
<tr>
<th>Table 1</th>
<th>Kinetics parameters and temperature feedback coefficient ($a_t$) for the MNSR Core.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Prompt Neutron Lifetime, $\mu$s</td>
<td>74.6</td>
</tr>
<tr>
<td>$\beta_{Dn}$, %</td>
<td>0.7826</td>
</tr>
<tr>
<td>Temperature, K</td>
<td>$x_0$, $\degree$C</td>
</tr>
<tr>
<td>293</td>
<td>0.00</td>
</tr>
<tr>
<td>313</td>
<td>-0.00722</td>
</tr>
<tr>
<td>333</td>
<td>-0.00698</td>
</tr>
<tr>
<td>353</td>
<td>-0.00802</td>
</tr>
<tr>
<td>373</td>
<td>-0.00803</td>
</tr>
</tbody>
</table>
inlet orifice by natural convection. The water flows past the hot fuel elements and comes out through the core outlet orifice. The hot water rises to mix with the large volume of water in the reactor vessel and to the cooling coil. Heat passes through the walls of the container to the pool water. The primary system represents the reactor vessel including reactor core and reactor primary components. The secondary system represents the reactor pool. The reactor was divided into a number of sections so that the division took into account all the components of the reactor and if there is any interchange between the components the model should be able to simulate this interchange. The objects G1–G10 represent coolant channels inside the core. Fig. 4 shows RELAP5 nodalization of MNSR. The annular beryllium, bottom beryllium and reactor vessel considered as the heat structure outside the core. The total number of control volume is 184.

### 2.3.2. Heat transfer model and correlations [10].

#### • Single-phase natural convection model:

If the wall (sheath) temperature remains below the liquid saturation temperature (in our case, 112 °C), the single-phase coefficient $h_{cai}$ must be determined. The Churchill and Chu correlation [10] can be used to find the wall-averaged Nusselt number for the entire Rayleigh number range (laminar, transition, and turbulent):

$$\text{Nu}_L = \left\{ \frac{0.825 + 0.387 \text{Ra}_L^{\frac{1}{4}}}{1 + (0.492 / \text{Pr})^{\frac{1}{3}}} \right\}^2$$  \hfill (1)

Where: $\text{Ra}_L$ - Rayleigh number, $\text{Pr}$ - Prandtl number, $\text{Gr}_L$ - Grashof number.

#### • Saturated Nucleate Boiling and subcooled nucleate boiling:

The Chen correlation is used for saturated and subcooled nucleate boiling. Although the correlation was based on saturated liquid conditions, it is used for subcooled liquid conditions by using the bulk liquid temperature as the reference temperature for the convective part of the correlation.

#### • Saturated nucleate boiling model basis:

The nucleate boiling correlation proposed by Chen has a macroscopic convection term plus a microscopic boiling term:

$$q'' = h_{\text{max}} (T_W - T_{\text{sat}}) F + h_{\text{mic}} (T_W - T_{\text{sat}}) S$$  \hfill (4)

Where: $T_W$–wall temperature. 

Chen chose Dittus-Boelter times a Reynolds number factor, $F$, for the convection part and Forster-Zuber pool boiling times a
suppression factor, $S$, for the boiling part. Between a liquid subcooling of zero and 5 K, the Chen F factor is linearly modified from the correlation value to 1.0, as follows:

$$F_0 = \frac{F_1}{C_0} \left( \frac{T_{spt}}{C_0} - \frac{T_f}{C_1} \right)$$

Where:

- $T_f$ = liquid temperature,
- $T_{spt}$ = steam saturation temperature based on total pressure.

### Subcooled nucleate boiling model basis

The subcooled boiling model was developed to generate bubbles in the superheated liquid next to the wall. A special model was needed because RELAP5 can only track the bulk liquid temperature. Actually, there is a superheated liquid layer next to the hot wall that is a source of steam. The model basis is the same as for saturated nucleate boiling expressed by Equation (4), with changes proposed by Bjornard and Griffith set F to one and use the total mass flux in the Reynolds number.

### Interfacial heat transfer (bubbly subcooled liquid) [10]

Interface heat transfer coefficient for subcooled bubbly flow regime is calculated by Unal correlation:

$$h_{if} = \frac{c \rho_l h_{fg} d}{2 \left( \frac{1}{p_s} - \frac{1}{p} \right)}$$

Where:

- $h_{fg}$-saturated enthalpy difference between steam and liquid.

$$\varphi = \begin{cases} 
1 & \nu_f \leq 0.61 \text{ m/s} \\
\nu_f & \varphi = 0.61 \text{ m/s} \\
0.47 & \nu_f > 0.61 \text{ m/s}
\end{cases}$$

$$C = \begin{cases} 
65 - 5.69 \cdot 10^{-5} (p - 10^{-5}) & 10^5 < p < 10^6 \text{ Pa} \\
0.25 \cdot 10^{10} \cdot p^{-1.418} & 10^6 < p < 17.7 \cdot 10^6 \text{ Pa}
\end{cases}$$

Where: $h_{if}$-saturated enthalpy difference between steam and liquid.

---

**Fig. 4.** Nodalization of the MNSR reactor using RELAP code.
Net vapor generation point (NVG) \[10,11\]

After the point of first bubble departure (also NVG point), significant void is present in the subcooled liquid and the void rises very rapidly even though the bulk liquid may still be in a highly subcooled state. It is assumed in RELAP5 that there are no bubbles until the NVG point in the flow channel, and the position of NVG is predicted using Saha-Zuber correlation as follows:

\[
\begin{align*}
    h_{\text{crit}} &= \begin{cases} 
    \frac{S \cdot T \cdot \rho_f}{0.0065} & \text{if } Pe > 7000 \\
    \frac{h_{f,sat}}{N \cdot u \cdot C_p} & \text{if } Pe \leq 7000 
    \end{cases} \\
    St &= \frac{N \cdot u \cdot C_p}{D \cdot k_f} \\
    Pe &= \frac{G \cdot D \cdot C_p}{k_f}
\end{align*}
\]

Where: \(q_f\) = wall heat flux to the liquid, \(h_{\text{crit}}\) is a “critical” enthalpy, \(h_{f,sat}\) is the saturated liquid enthalpy, \(D\) is the hydraulic diameter, and \(G\) is the total mass flow rate.

If \(h_{f,sat} > h_{\text{crit}}\), then boiling occurs. And the fraction of wall heat flux used for liquid evaporation \(\chi\) is calculated as follows:

\[
\chi = \frac{\min(h_f, h_{f,sat}) - h_{\text{crit}}}{h_{f,sat} - h_{\text{crit}}(1 + \varepsilon)}
\]

Where \(\varepsilon\) is calculated using Lahey model:

\[
\varepsilon = \frac{\rho_l}{\rho_g} \left[ 1 - \frac{h_{f,sat} - h_f}{h_{f,sat}} \right]
\]

Where: \(\rho_l\) and \(\rho_g\) is liquid and vapor density respectively and \(h_{fg}\) is the latent heat.

The vapor generation in wall region \(\Gamma_w\) is

\[
\Gamma_w = \frac{q_f^w \cdot A_w}{V \cdot h_{fg}} \chi
\]

Where: \(A_w\) is heat transfer area, and \(V\) is volume.

When bubbles generated in near wall region detach from the wall into the bulk, condensation in the bulk occurs. For bubbly flow, the modified Unal-Lahey correlation is used to calculate the interfacial heat transfer coefficient. The interfacial heat transfer area is as follows:

\[
A_{gf} = \frac{3.6 \cdot \varepsilon_{fg}}{d_0}
\]

Where: \(\varepsilon_{fg}\) is void fraction, and \(d_0\) is the mean bubble diameter.

3. Simulation results and discussion

The results have been conducted without a gap between the fuel and clad, but the sensitivity studies on the gap size of fuel model can be seen in Fig. 14.

Figs. 5 and 6 compare experimental measurements and calculated results for the 3.6 mk reactivity insertion with HEU fuel. The calculated results agree almost exactly with the measured data. This shows that the RELAP thermal-hydraulic model and the reactivity feedback coefficients accurately model the MNSR reactor with HEU fuel. The laminar flow was changed to turbulent flow due to the achieved high power in the core. Fig. 5 shows the power is increased to a level of 3.44 of the nominal power after 240 s. After that, the power decreases due to the reactivity negative feedback effect (temperature effect and xenon accumulation) \[7\]. The RIA was simulated to demonstrate the inherently safe power excursion in case of full withdrawal of the control rod. Fig. 6 shows the core temperature is increased to a level of 2.25 of the initial core temperature.

Fig. 7 shows the reactor power for the three RIA. During the reactivity insertion the reactor goes on to a very short period, and it exhibits an initial large power spike -that lasts substantially less than a second-before it drops after the insertion ends. The power first went up and then dropped down gradually due to the negative temperature feedback reactivity. Fig. 7 shows the power peaks were: 103 kW at 240 s, 174 kW at 160 s and 195 kW at 140 s after the reactivity insertion for the cases of 3.6, 6.0, and 6.53 mk, respectively.
Fig. 8 shows the reactivity behavior vs. the time. Initially, the reactivity is increased to the maximum value (first peak), after that the value is decreased by the temperature negative feedback. The reactivity increase (secondary peak) is caused by the slowly increasing coolant velocity effecting its heat removal capabilities.

The impact of prompt neutron behavior on the evolutions of fuel, clad and coolant temperatures is presented exemplary in Figs. 9–11 for the cases of 3.6, 6.0, and 6.53 mk, respectively. As expected the fuel temperature follows strictly the power behavior and reaches its maximum of 116 °C at 240 s, 124 °C at 160 s and 126 °C at 140 s respectively. Due to the good thermal conductivity of clad alloy the clad temperature reaches its maximum of 115, 122 and 123 °C at almost the same time of fuel temperature.

The development of mass flow rate resulting from the natural circulation during various RIA is presented in Fig. 12. One can notice that the mass flow rates after 200 s were 0.0115, 0.0141 and 0.0147 kg/s for RIA of 3.6, 6.0 and 6.53 mk respectively.

Fig. 13 shows the development of flow velocity of coolant for various reactivity insertions. It is noticed the increase in average core velocity during RIA. The velocity after 200 s were: 17.5 mm/s (compared to 12 mm/s at nominal power), 22.1 and 23.1 mm/s for RIA of 3.6, 6.0 and 6.53 mk respectively. The ATHLET code gave the same result [12].

The transient behavior of the coolant temperature is similar to power behavior. The maximum core outlet temperatures were reached about 65.5, 81.5 and 85.6 °C for RIA of 3.6, 6.0 and 6.53 mk, respectively. In all these cases, the saturation temperature was not exceeded indicating that saturated boiling did not exist (subcooled boiling was observed for the three considered RIA). The maximum core inlet temperatures were reached about 31.0, 36.0 and 38.0 °C for RIA of 3.6, 6.0 and 6.53 mk, respectively.

The gap conductance between the fuel and the cladding depends strongly on the gap width and has a significant influence on the fuel temperatures. In order to assess the impact of a variation in the gap between the fuel rods and the cladding, 3.6 mk reactivity insertion cases were run with gap sizes of both 50 μm and 100 μm.

Fig. 14 shows the results of these analyses. As expected, a larger gap size results in a higher fuel temperature, and the peak power is an unchanged (The Doppler reactivity feedback is very small in the HEU fuel—the least amount of U-238).

Table 2 summaries the comparison of the RELAP5 code results against ATHLET [12] for RIA of 3.6 mk. As it is noticed, the RELAP5 results were more accurate than the ATHLET result compared with experimental values, due to the approximation made in the ATHLET code (six fuel instead of ten groups).

Table 3 shows the comparison of the RELAP5 code results against MERSAT [13] and PARET/ANL [14] codes for RIA of 5.5 mk. The discrepancy was observed between results. The difference between the PARET/ANL and the RELAP5 results are related to the approximation in the PARET/ANL where two channels were only considered.

Table 4 shows the power peak, fuel and clad temperature peaks for various reactivity insertions. The power peaks were: 103, 174 and 195 kW, respectively. The fuel temperature peaks were: 116, 124 and 126 °C, respectively.

4. Conclusion

RIAs of 3.6, 6.0, and 6.53 mk were studied in the MNSR using the RELAP5 code. The maximum reactor powers were 103, 174 and 195 kW, respectively. The maximum fuel temperatures were 116, 124 and 126 °C, respectively. The maximum clad temperatures were 115, 122 and 123 °C, respectively. Good agreement was noticed between our results, measured and previously published results. Subcooled boiling was observed for the three considered RIA (Tr < Tsat = 112 °C and Tw > Tgas). These results verify the safety of the MNSR under all the possible RIAs. The MNSR operation limits during the RIA remain within the design limits due to the high negative temperature coefficient of the moderator.
Fig. 12. Mass flow rate from the natural circulation during various RIA.

Fig. 13. Variation of flow velocity of coolant for various reactivity insertions.

Fig. 14. Fuel temperature profiles as a function of gap thickness.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Code AHTLET [12]</th>
<th>RELAP5/Mod3.2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Peak power (kW)</td>
<td>99.0</td>
<td>103.0</td>
</tr>
<tr>
<td>Peak clad temperature (°C)</td>
<td>115.5</td>
<td>115.0</td>
</tr>
<tr>
<td>Peak fuel temperature (°C)</td>
<td>119.0</td>
<td>116.0</td>
</tr>
<tr>
<td>Outlet temperature (°C)</td>
<td>62.5</td>
<td>65.5</td>
</tr>
</tbody>
</table>

Table 2
Comparison of the code results for RIA of 3.6 mk (RELAP5 against AHTLET).

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</thead>
<tbody>
<tr>
<td>Peak power (kW)</td>
<td>163.04</td>
<td>146.8</td>
<td>140.0</td>
</tr>
<tr>
<td>Peak clad temperature (°C)</td>
<td>112.36</td>
<td>125.6</td>
<td>119.0</td>
</tr>
<tr>
<td>Peak fuel temperature (°C)</td>
<td>160</td>
<td>142.4</td>
<td>121.0</td>
</tr>
<tr>
<td>Outlet temperature (°C)</td>
<td>70</td>
<td>69.0</td>
<td>74.0</td>
</tr>
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</table>

Table 3
Comparison of code results for RIA of 5.5 mk (RELAP5 against MERSAT and PARET/ANL).
Table 4

<table>
<thead>
<tr>
<th>Reactivity insertion mk</th>
<th>power peak kW</th>
<th>fuel temperature Peak °C</th>
<th>clad temperature Peak °C</th>
</tr>
</thead>
<tbody>
<tr>
<td>3.6</td>
<td>103</td>
<td>116</td>
<td>115</td>
</tr>
<tr>
<td>6.0</td>
<td>174</td>
<td>124</td>
<td>122</td>
</tr>
<tr>
<td>6.53</td>
<td>195</td>
<td>126</td>
<td>123</td>
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</tbody>
</table>

Declarations of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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Appendix A. Supplementary data

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References